NMC
Committed to Nuclear Excellence

Nuclear Management Company, LLC
Prairie Island Nuclear Generating Plant
1717 Wakonade Dr. East • Welch MN 55089

October 2, 2001

10 CFR Part 50
Section 50.73

U S Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
Docket No. 50-282 License Nos. DPR-42

LER 1-01-05:
Fault and Fire in Non-Safeguards Circuit Breaker Results in Reactor Trip and Auxiliary Feedwater System Actuation

The Licensee Event Report for this occurrence is attached. In the report, we made several new NRC commitments identified as the corrective actions in ***italics***. This event was reported via the Emergency Notification System in accordance with 10 CFR Part 50, Section 50.72, on August 3, 2001. Please contact us if you require additional information related to this event.

Mano Nazar
Site Vice President
Prairie Island Nuclear Generating Plant

c: Regional Administrator - Region III, NRC
NRR Project Manager, NRC
Senior Resident Inspector, NRC
James Bernstein, State of Minnesota

Attachment

IE22

NRC FORM 366 (1-2001) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 6-30-2001
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory information collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to bjs1@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Prairie Island Nuclear Generating Plant Unit 1	05000 282	1 OF 5

TITLE (4)
Fault and Fire in Non-Safeguards Circuit Breaker Results in Reactor Trip

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
									Prairie Island Unit 2	05000 306
08	03	01	01	– 05 –	00	10	02	01	FACILITY NAME	DOCKET NUMBER

OPERATING MODE (9)	1	THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply) (11)							
			20.2201(b)		20.2203(a)(3)(ii)		50.73(a)(2)(ii)(B)		50.73(a)(2)(ix)(A)
POWER LEVEL (10)	≈25		20.2201(d)		20.2203(a)(4)		50.73(a)(2)(iii)		50.73(a)(2)(x)
			20.2203(a)(1)		50.36(c)(1)(i)(A)	√	50.73(a)(2)(iv)(A)		73.71(a)(4)
			20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)		50.73(a)(2)(v)(A)		73.71(a)(5)
			[redacted]		[redacted] 50.46[redacted]		[redacted]		OTHER Specify in Abstract below or in NRC Form 366A
			20.2203(a)(3)(i)		50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
Jeff Kivi	651-388-1121

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX

SUPPLEMENTAL REPORT EXPECTED (14)				EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE).	√	NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On August 3, 2001, with Unit 2 at 100% power, Unit 1 was undergoing an orderly startup following the reactor trip that occurred on August 1, 2001. At about 8:40 pm CDT, with Unit 1 at approximately 25% power, operators were transferring the power supply to the non-safeguards 4 kV buses from the Unit 1 Reserve transformer to the Unit 1 Main Auxiliary transformer. Within seconds of closing the source breaker from the Main auxiliary transformer to 4 kV Bus 12, the breaker failed. The breaker failure initiated a fire in Cubicle 12-4 of Bus 12. The failure actuated the Generator Transformer protective relaying scheme, including a lockout of Bus 12. The bus lockout opened the Breaker 12-1, Bus 12 source from the 1RX transformer, and actuated the protective relaying scheme. The protective relaying scheme caused a turbine/reactor trip and actuation of the Auxiliary Feedwater System.

To support firefighting, Bus 11 was deliberately de-energized, thus, both reactor coolant pumps were unpowered and the unit was cooled down via natural circulation. The fire was extinguished about one hour and 30 minutes after the event. The unit was brought to cold shutdown within the time allowed by Technical Specifications. Remaining unit equipment responded as expected and, overall, the organization effectively responded to the event (although some improvements were recommended following an evaluation of organizational response).

A root cause evaluation is complete, the damaged equipment has been repaired, and actions are being taken to prevent recurrence of the event and to improve organizational response to events.

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
Prairie Island Nuclear Generating Plant Unit 1	05000 282	01	05	00	2 OF 5

NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

EVENT DESCRIPTION

On August 3, 2001, with Unit 2 at 100% power, Unit 1 was undergoing an orderly startup following the reactor trip that occurred on August 1, 2001. At about 8:40 pm CDT, with Unit 1 at approximately 25% power, operators were transferring the power supply to the 4kV non-safeguards buses from the Unit 1 Reserve (1R) transformer to the Unit 1 Main Auxiliary (1M) transformer. Within seconds of closing Breaker 12-4, 4 kV Bus 12 source from 1MY transformer, the breaker failed. The breaker failure initiated a fire in cubicle 12-4 of Bus 12. The failure actuated the Generator Transformer protective relaying scheme, including a lockout of Bus 12. The bus lockout opened Breaker 12-1, Bus 12 source from the 1RX transformer, and actuated the protective relaying scheme. The protective relaying scheme caused a turbine/reactor trip and actuation of the Auxiliary Feedwater System.

The lockout of Bus 12, which supplies the 12 Main Feedwater Pump (MFWP - not running at the time) and the 12 Reactor Coolant Pump (RCP), left the 12 RCP unpowered. During the course of firefighting, operators de-energized Bus 11, as part of de-energizing 1R Transformer. Consequently, power to 11 RCP was lost and the reactor was subsequently cooled by natural circulation (in accordance with Technical Specification 3.1.A.b(3)). Approximately one and one half hours after the initial breaker failure, the Prairie Island Fire Brigade (with assistance from the Red Wing Fire Department) extinguished the fire.

In addition to the fire, which resulted in a Notification of Unusual Event (NUE) and the reactor trip, the 11 and 12 Auxiliary Feedwater Pumps automatically started on lo-lo steam generator level, as would be expected for a reactor trip.

During the fire, two employees were treated on site for heat exhaustion. One of these employees was transported to the hospital for heat exhaustion. These employees were members of the fire brigade. Upon extinguishing the fire, stabilizing the plant, and verifying all safeguards buses were powered from offsite sources, plant personnel terminated Unusual Event classification at approximately 12:10 am CDT on August 4, 2001. Technical Specification required staffing levels were maintained.

CAUSE OF THE EVENT

A root cause investigation was conducted for the breaker failure. The investigation team used a Failure Modes and Effects Analysis process to determine the most probable cause of the event. The team concluded the root cause of the event was a poor electrical connection between the Breaker 12-4 C-phase primary disconnect assembly (PDA) and the 1MY bus stab, which led to overheating of the PDA, which in turn led to a failure of the PDA one or two seconds after Breaker 12-4 was closed. The failure of the PDA led to a C-phase to ground arcing event, which quickly involved all phases. The arcing led to actuation of the protective relaying, which resulted in a turbine/reactor trip.

The poor electrical connection was caused by poor conductive surfaces. A thinned silver coating on the connections exposed copper oxides in the conductors. Because copper oxide has resistance that is orders of magnitude higher than copper, silver, or silver oxide, this led to a high resistance connection, and overheating. The silver layer was most likely thinned by environmental and operating conditions and/or maintenance practices.

ANALYSIS OF THE EVENT

This event is reportable per 10CFR 50.73(a)(2)(iv)(A) because it resulted in a reactor trip and because it resulted in the automatic actuation of Auxiliary Feedwater.

Equipment and Organizational Response to Event

With the exception of the initiating component failure, all equipment response was as expected. Both reactor coolant pumps (RCPs) lost power, but 12 RCP lost power because the event started in Bus 12 and 11 RCP lost power because Bus 11 was deliberately de-energized to support firefighting efforts on Bus 12.

Overall, the organization's response to the event was effective. The Fire Brigade effectively responded and extinguished the fire. The natural circulation cooldown was performed without incident. The Emergency Response Organization (ERO) was partially activated as a conservative measure to provide additional resources. ERO activation is not required for a NUE. The post-activation review identified that there is little guidance for partial activation in response to a NUE.

Additionally, post-activation review identified some challenges due to on-shift staffing levels. Fire fighting personnel were used to perform electrical switching while in full suitup and this contributed to two members suffering heat exhaustion. The Shift Manager may have had difficulty fulfilling his role as Shift Technical Advisor and interim Emergency Director had it not been for additional support from ERO personnel already on site. The only Emergency Medical Technician (EMT) onsite at the time of the fire was also the Shift Emergency Communicator (SEC), which could have delayed offsite notifications if the event timeline had been slightly different.

Extent of Condition/Potential for Common Mode Failure Mechanism

The Root Cause Evaluation determined that maintenance practices (use of abrasives to clean PDAs and bus stabs and use of silver plating substitutes in some instances) could have contributed to the failure of the PDA. This cannot be conclusively stated because any evidence of this would have been effectively destroyed during the event. The Root Cause Evaluation also notes that environmental and operating conditions could have contributed to the failure as well. Inspections have been conducted on a number of other breakers.

To date, over 25% of the approximately 100 breakers of this type inservice on site have been inspected. Inspections focused on source breakers, since operating conditions are suspected to be a contributor to the failure of breaker 12-4. Six source breakers on Unit 1 safeguards buses 15 and 16 have been inspected. 25 Unit 2 safeguards 4 kV breakers were newly installed as part of the modifications resulting from implementation of the Station Blackout Rule and are expected to be less susceptible to this failure mode. All the breakers in nonsafeguards buses 11 and 12 have been inspected (except the failed breaker 12-4) and sixteen breakers in nonsafeguards buses 13 and 14 have been inspected. Thus far, one other breaker has been found with signs of heating in the PDA. This breaker is 11-4, the 1MY supply to 4kV Bus 11, i.e., it is similar to breaker 12-4.

Since maintenance practices are common to all breakers on the site, the potential for a common mode failure has been assessed. Records of past maintenance on 4kV breakers does not conclusively note whether the suspect maintenance practices may have been performed. Thus, operability evaluations have been completed generically for Unit 1 and Unit 2 safeguards 4kV breakers. The operability evaluations note that the breakers in the safeguards buses see much less load than Breaker 12-4. The operability evaluations also note that safeguards buses get periodic maintenance to ensure operability and periodic surveillance testing to demonstrate operability.

Performance Indicator Impact

This event will contribute to the performance indicators for reactor scrams and reactor scrams with loss of normal heat removal. No safety functions were lost as a result of this event and no trains of any of the four safety systems incurred any unavailability as a result of this event. Performance indicators related to emergency preparedness, radiation protection and physical protection do not apply.

Risk Significance Determination

Both Incremental Core Damage Probability (ICDP) and Conditional Core Damage Probability (CCDP) related to the Bus 12 fire event were calculated. For an ICDP calculation, the focus is on configuration risk assessment. In this case, all the initiating events in the PRA model are included and the potential risk during the period of natural circulation was calculated. On the other hand, the focus of a CCDP calculation is on event risk assessment under certain given conditions. In this case, a successful reactor trip is the given condition and the risk assessment is mainly based on the responses of the plant safety systems to the event (no other initiating events are assumed to occur after the successful reactor trip).

The risk of operating Unit 1 during the 54 hours of natural circulation, ICDP, and the event (Bus 12 fire) itself, CCDP, were assessed. Since both calculated ICDP and CCDP were low (1.40E-7 and 8.88E-8 respectively), this event is considered to be non-risk significant.

CORRECTIVE ACTION

Immediate Actions

1. All bus to breaker stabs in Bus 11 and 12 were re-silvered and all the PDAs for breakers in those buses were replaced.

2. In addition, special inspections were performed on selected source breakers for buses 11, 13, 14, 15, and 16 before Unit 1 was placed online again.

NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

Planned Actions

3. ***In order to determine the extent of the condition, visual inspection of PDAs is to be performed when a breaker is racked to the disconnect position.*** This action will be maintained until all 4 kV of this type have been inspected and maintenance procedures are revised.

4. ***Breaker maintenance procedures will be changed to require documentation of the as-found and as-left condition of the stabs and PDAs and to require that cognizant system engineers review the resulting work packages.***

5. ***Detailed procedures will be developed to govern the use and application of substitute silver coating substances.***

6. ***The practice of cleaning stabs and then returning them to service will be reviewed.***

7. ***Tests or inspections will be developed that verify the adequacy of primary connections in 4 kV switchgear.***

8. Process for transition to Recovery phase following emergency plan activation will be improved.

9. Evaluate the Event Response Team recommendation to eliminate the practice of partial ERO activation will be eliminated.

10. The adequacy of on-shift staffing (specifically with respect to potentially overlapping duties of Shift Manager, STA, fire brigade, SEC, and EMT) required for dealing with plant events will be reassessed.

FAILED COMPONENT IDENTIFICATION

The breaker in cubicle 12-4 was an ABB 5HK250, serial #44477J2110C. The failure was attributed to a failed PDA.

PREVIOUS SIMILAR EVENTS

No other examples of catastrophic breaker failure have occurred at Prairie Island in the past three years.